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Proprietary – Withhold under 10 CFR 2.390. Enclosure 2 contains PROPRIETARY information.

October 22, 2020
GO2-20-127

10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO FOURTH TEN-YEAR INTERVAL INSERVICE INSPECTION (ISI) PROGRAM RELIEF REQUEST 4ISI-09**

References:
1. Letter from J Kent Dittmer, Energy Northwest, to NRC, “Fourth Ten-Year Interval Inservice Inspection (ISI) Program Relief Request 4ISI-09,” dated April 22, 2020 (ADAMS Accession Number ML20114E235 and ML20114E236)
2. Email from M. Chawla, NRC, to Rick Garcia, Energy Northwest, “Columbia Generating Station - Final - Request for Additional Information - Fourth Ten-Year Interval Inservice Inspection (ISI) Program Relief Request 4ISI-09 - EPID L-2020-LLR-0068,” dated September 23, 2020 (ADAMS Accession Number ML20267A516)

Dear Sir or Madam:

By Reference 1 Energy Northwest submitted a relief request to change the number of Reactor Pressure Vessel Feedwater nozzle examinations from 100% to 25% per inservice inspection interval. By Reference 2 the Nuclear Regulatory Commission (NRC) requested additional information related to the Energy Northwest submittal. Enclosure 1 to this letter contains the requested information.

Enclosure 2 to this letter contains corrected page 13 of 1801567.301P, Revision 1. Structural Integrity Associates considers certain information contained in Enclosure 2 to be proprietary and, therefore, requests that it be withheld from public disclosure in accordance with 10 CFR 2.390. Enclosure 3 of this letter contains corrected page 13 for the non-proprietary version 1801567.301, Revision 1. Enclosure 4 contains the associated affidavit for the request to be withheld from public disclosure.

When Enclosure 2 is removed from this letter, the letter and remaining Enclosures are NON-PROPRIETARY.

There are no new commitments made in this submittal. If you have any questions or require additional information, please contact Mr. R. M. Garcia, Licensing Supervisor, at 509-377-8463.

Executed this 22 day of October, 2020.

Respectfully,

DocuSigned by:
J. Kent Dittmer
60D4E7CA44C64D4...

J. Kent Dittmer
Vice President Engineering

Enclosures: As stated

cc: NRC RIV Regional Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector -988C
CD Sonoda – BPA - 1399
EFSECutc.wa.gov – EFSEC
E Fordham – WDOH
R Brice – WDOH
L Albin – WDOH

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

<u>Background</u>

By letter dated April 22, 2020, (Agencywide Documents and Access Management System (ADAMS) Accession No. ML20114E234), Energy Northwest (the licensee) requested relief from certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1 for the inservice inspection (ISI) program at the Columbia Generating Station (CGS).

Pursuant to 10 CFR 50.55a(z)(1), the licensee submitted Relief Request 4ISI-09 for the performance of alternate examinations of the reactor vessel feedwater nozzle-to-shell welds and feedwater nozzle inner radii on the basis that the proposed alternative would provide an acceptable level of quality and safety.

<u>Regulatory Basis</u>

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), which states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI.

The regulations in 10 CFR 50.55a(g) require that the ISI of ASME Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code and applicable addenda. The ASME Code, Section XI, requires that all reactor vessel nozzles to be inspected during each 10-year ISI interval. The volumes in each nozzle required to be inspected are 100 percent of the nozzle-to-vessel shell weld volume and 100 percent of the nozzle inner radius section volume, as shown in the applicable figure in Figures IWB-2500-7(a) through (d) "Nozzle in Shell or Head," of the ASME Code, Section XI.

<u>Request for Additional Information</u>

To complete its review, the Nuclear Regulatory Commission (NRC) requests the following additional information.

1. Inspection

NRC REQUEST NO. 1.1:

Issue:

The NRC staff noted that the relief request does not discuss the inspection strategy if an indication is detected in a feedwater nozzle or in a nozzle-to-shell weld in the future, whether an expansion (extent of condition) inspection will be performed.

Request:

Discuss the expansion (extent of condition) inspection if an indication is detected in a feedwater nozzle or in a nozzle-to-shell weld in the future. If no expansion inspection will be performed, provide justification.

ENERGY NORTHWEST RESPONSE TO RAI 1.1:

This request does not seek relief from any other aspect of the ASME Section XI Code. Therefore, if an indication is detected that exceeds ASME inspection criteria, scope expansion (extent of condition) will be performed in accordance with ASME Section XI, subsection IWB-2430 “Additional Examinations”, for the Code of Record in place at the time of discovery.

NRC REQUEST NO. 1.2:

Issue:

On Page 5 of the relief request, the licensee states that it reviewed the most recent examination results for the subject components and reported that no recordable indications in the feedwater nozzle inner radii or nozzle-to-shell welds. The licensee further stated that all the examinations had greater than 99% examination coverage. It is not clear to the NRC staff whether all six feedwater nozzles were examined, what examination method(s) were used, and whether the 99% examination coverage is applicable to all the six feedwater nozzles and associated welds.

Request:

a) Confirm that all six-feedwater nozzle inner radii and associated nozzle-to-shell welds were inspected in the most recent examination.
b) Discuss any other examination method used to inspect the subject components besides the ultrasonic testing.
c) Discuss whether the 99% examination coverage is applicable to all six feedwater nozzle radii and nozzle-to-shell welds.

ENERGY NORTHWEST RESPONSE TO RAI 1.2:

a) Table A below provides the year of the most recent examination to confirm that all six-feedwater nozzle inner radii and all six feedwater nozzle-to-shell welds have been inspected.

Table A

Identification Number	Description	Year of Last Exam	Exam Method	Percent Coverage
N4-30	RFW Nozzle-to-shell Weld @ 30 Degrees	2019	UT	99.4%
N4-30-IR	RFW Nozzle Inner Radius @ 30 Degrees	2019	UT	100%
N4-90	RFW Nozzle-to-shell Weld @ 90 Degrees	2019	UT	99.4%
N4-90-IR	RFW Nozzle-to-shell Weld @ 90 Degrees	2019	UT	100%
N4-150	RFW Nozzle-to-shell Weld @ 150 Degrees	2015	UT	99.1%
N4-150-IR	RFW Nozzle Inner Radius @ 150 Degrees	2015	UT	100%
N4-210	RFW Nozzle-to-shell Weld @ 210 Degrees	2015	UT	99.1%
N4-210-IR	RFW Nozzle Inner Radius @ 210 Degrees	2015	UT	100%
N4-270	RFW Nozzle-to-shell Weld @ 270 Degrees	2015	UT	99.1%
N4-270-IR	RFW Nozzle Inner Radius @ 270 Degrees	2015	UT	100%
N4-330	RFW Nozzle-to-shell Weld @ 330 Degrees	2015	UT	99.1%
N4-330-IR	RFW Nozzle Inner Radius @ 330 Degrees	2015	UT	100%

RFW = Reactor Feedwater
UT = Ultrasonic Testing

b) Only the volumetric examinations required by ASME Section XI, Table IWB-2500-1, Category B-D are discussed in this relief request. As reported in Table A, the volumetric examination method used for each examination was ultrasonic testing (UT).

c) Table A above provides the percent coverage for each examination. As seen in the table, greater than 99% coverage was achieved for all six feedwater nozzle radii and all six feedwater nozzle-to-shell welds.

2. Deterministic Stress Analysis

NRC REQUEST NO. 2.1:

Issue:

First paragraph on Page 5 of the relief request states that the cladding on the feedwater nozzle inner radius has been removed. However, Enclosure 2 to the licensee's April 22, 2020 letter does not specifically mention that cladding is not modeled on the nozzle radius in the finite element analysis. Also, the relief request does not include a drawing to show location of the cladding on the reactor vessel shell that near the nozzle inner radius. The NRC staff notes that the location of cladding is significant because thermal expansion of cladding is different from that of the reactor vessel shell.

Request:

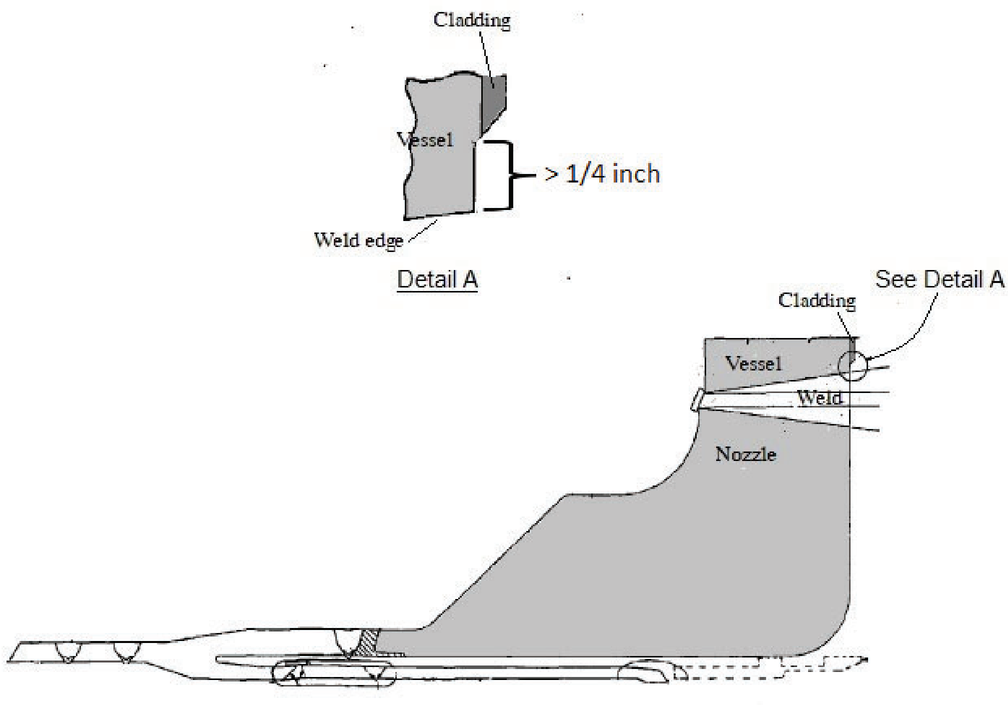
(a) Confirm that the finite element model in Enclosure 2, Figure 3 does not include cladding on the inner surfaces of feedwater nozzles and associated nozzle-to-shell welds. If there is no cladding on the nozzle-to-shell welds, justify that the clad stress due to the thermal expansion difference between the clad and the reactor vessel shell of the adjacent cladded inner surface shown in Figure 3 has negligible impact on stresses in the nozzle-to-shell welds.
(b) Provide a sketch to show the distance from the feedwater nozzle or nozzle-to-shell weld to the reactor vessel shell that has no cladding.
(c) Confirm that the finite element model of the feedwater nozzle radius and nozzle-to-shell weld is consistent with the actual field configuration.

ENERGY NORTHWEST RESPONSE TO RAI 2.1:

a) It is confirmed that the finite element model in Enclosure 2, Figure 3 did not include cladding on the inner surfaces of feedwater nozzles and associated nozzle-to-shell welds. In Figure 3 of Enclosure 2, the leftmost figure shows the cladding as a thin blue layer relative on the inside surface of the RPV base metal, which is shown in red. Because the cladding was explicitly included in the model, the clad stress due to the thermal expansion difference between the cladding material and the reactor vessel shell material was accounted for in the finite element analysis.

b) Figure A below provides a sketch based on the N4 Nozzle Assembly drawing, 02B13-06,141 Revision 10, the distance from the outside edge of the nozzle-to-shell weld to the edge of the cladding is greater than ¼ inch.

Figure A
Sketch of N4 Feedwater Nozzle Assembly



c) It is confirmed that the finite element model of the feedwater nozzle radius and nozzle-to-shell weld was consistent with the actual field configuration, as shown in the sketch in RAI2.1(b) above.

NRC REQUEST NO. 2.2:

Issue:

The reactor vessel nozzles analyzed in BWRVP-108-A, “BWR Vessel and Internals Project: Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,” and BWRVIP-241-A, “BWR Vessel and Internals Project Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,” are represented as a 360-degree nozzle configuration in the finite element model. However, the finite element model in Enclosure 2, Figure 3 shows a quarter of the feedwater nozzle. The NRC staff noted that the azimuthal locations (i.e., 0, 90, 180, or 270 degrees) of a feedwater nozzle may experience different stresses. In addition, the finite element model in Figure 3 shows that the reactor shell on which the feedwater is attached is also a quarter-size panel.

Request:

a) Discuss the adequacy of the quarter size feedwater nozzle model in Enclosure 2, Figure 3 to generate the appropriate stress distributions as compared to the 360-degree full nozzle model.
b) Clarify how would the stresses extracted from the quarter-sized finite element model in Enclosure 2, Figure 3 represent the appropriate stresses to calculate the probability of failure when the full 360-degree nozzle is not represented in the finite element model.
c) Clarify whether the quarter sized reactor shell modeled in Figure 3 will provide accurate stress distribution in the feedwater nozzle radius region and nozzle-to-shell welds.

ENERGY NORTHWEST RESPONSE TO RAI 2.2:

a) In the evaluation in Enclosure 2, the feedwater piping and nozzle were axisymmetric. The nozzle-to-shell weld had two axes of symmetry in the longitudinal and circumferential directions of the vessel shell. Hence, a quarter model (0º to 90º) with the appropriate boundary conditions was adequate for the stress analysis of loads that were axisymmetric and resulted in the same stresses as a 360º model. Hence, the axisymmetric pressure loading and thermal loadings were analyzed using the quarter size feedwater nozzle model in Enclosure 2, Figure 3.

   Although not evident in Enclosure 2, for nozzle moment loadings due to thermal expansion which were not axisymmetric, a separate full 3-D FEM (360° model) was constructed from the quarter size model by reflecting about the symmetry planes. Unit piping interface moments were applied at one free end of the pipe. Three independent, orthogonal unit moment loading cases (three separate load steps) were performed as follows, in one stress analysis for the evaluation in Enclosure 2:

   Load step 1: MX = 1,000 in-lb.
   Load step 2: MY = 1,000 in-lb.
   Load step 3: MZ = 1,000 in-lb.

b) The stresses for the quarter model for axisymmetric loads (pressure and thermal) were repeated in the other three quadrants. For instance, the stresses at 0º were identical to those at 180º, and the stresses at 90º were identical to those at 270º that makes the use of a full 360º model superfluous. The repeatable nature of the stress distributions can be observed in Figures 4-30 through 4-37 in BWRVIP-108-A where a 360º model was used.

c) Based upon responses to Item (a) and (b) above, it is clarified that the quarter sized reactor shell modeled in Figure 3 of Enclosure 2 will provide accurate stress distribution in the feedwater nozzle radius region and nozzle-to-shell welds for the axisymmetric loads (pressure and thermal).

NRC REQUEST NO. 2.3:

<u>Issue:</u>

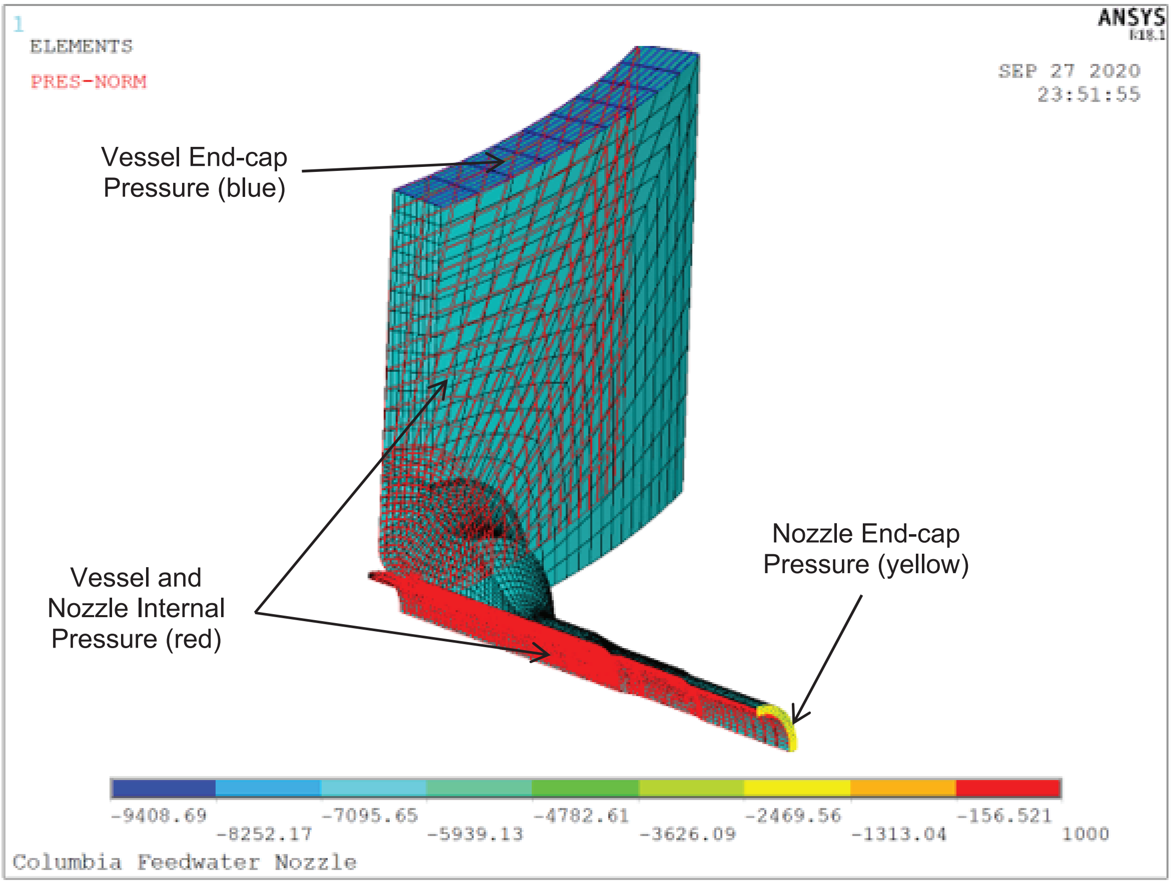
Enclosure 2, Figure 4 is labeled as the applied pressure and boundary conditions. However, the NRC staff is not clear exactly what are the applied pressure and boundary conditions in Figure 4.

<u>Request:</u>

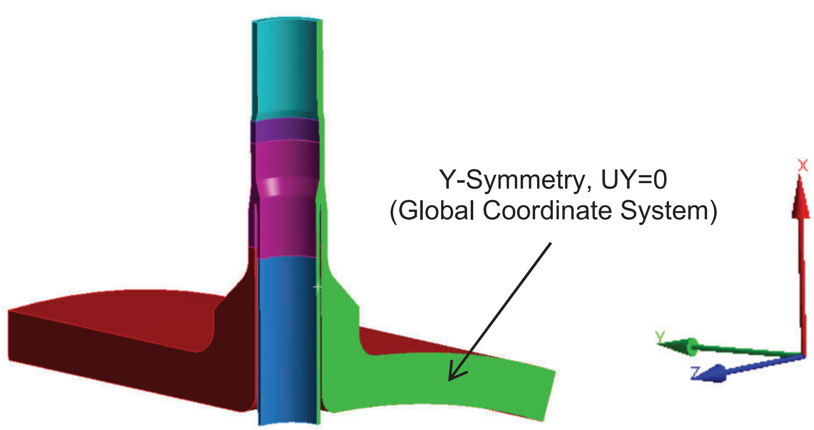
Clarify what are the pressure and boundary conditions that are applied to the finite element model as shown in Figure 4.

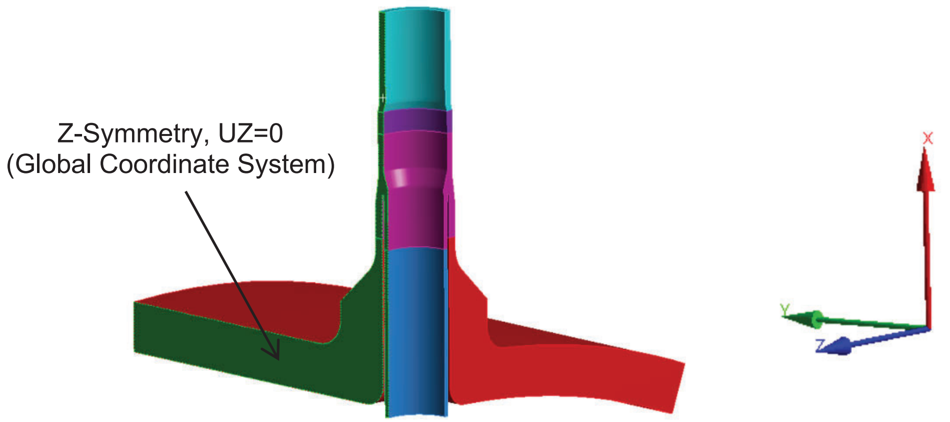
ENERGY NORTHWEST RESPONSE TO RAI 2.3:

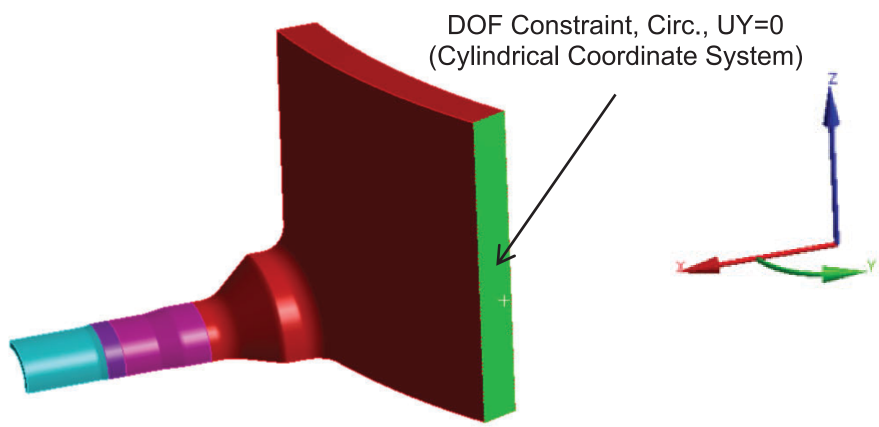
Using the finite element model from the evaluation, Figure 4 of Enclosure 2 has been replotted from a different perspective to more clearly show the applied internal pressure and end-cap pressures, as discussed in Section 4.1.2 of Enclosure 2l:



Using the finite element model from the evaluation, the following figures show the applied boundary conditions separately:







NRC REQUEST NO. 2.4:

<u>Issue:</u>

Section 3.0 in Enclosure 2 describes applied loadings. However, it appears that the applied loading from the feedwater pipe imposed on the feedwater nozzle was not included in the finite element analysis. The NRC staff notes that the loading from the feedwater pipe may cause stresses on the feedwater nozzle and, therefore, should be considered in the stress analysis of the feedwater nozzle.

Request:

Discuss whether the forces and moments generated from the feedwater pipe are included in the stress analysis of the feedwater nozzle and nozzle-to-shell weld in the finite element analysis. If not, provide justification.

ENERGY NORTHWEST RESPONSE TO RAI 2.4:

The nozzle moment due to thermal loads were included in the stress analyses. The thermal moments were calculated in Section 3.1.2 of Enclosure 2. Axial load was not included in the analysis since the stress due to the axial force has a negligible effect as compared to the stress due to the applied moments (less than 0.7% of the stresses due to the moments). This has been verified through a subsequent finite element analysis using the same finite element model in Enclosure 2.

NRC REQUEST NO. 2.5:

Issue:

Enclosure 2, Tables 1 to 4, provide transient definitions for various events. Enclosure 2 states that the thermal transient cycles were predicted for 60 years of operation and that it follows the methodology used in BWRVIP-108-A and BWRVIP-241. However, it is not clear to the NRC staff the exact source of the transient cycles and definitions in the stress analysis.

Request:

Discuss whether the thermal cycles and transients used in the stress analysis of the feedwater nozzle radius and nozzle-to-shell weld in Enclosure 2 come from the plant-specific licensing basis, or from the generic transients as shown in BWRVIP-108-A and BWRVIP-241. If the generic thermal cycles and transients in these BWRVIP reports were used, discuss whether they bound the plant-specific transient data at CGS.

ENERGY NORTHWEST RESPONSE TO RAI 2.5:

The stress analysis used plant specific transients in Tables 1 through 4 of Enclosure 2 from the CGS thermal cycle diagrams.

NRC REQUEST NO. 2.6:

Issue:

Enclosure 2, Section 3.1.2 states that only cyclic loads such as thermal transient and pressure are included in the stress analysis and that deadweight, which does not cycle, is not needed. The NRC noted that non-cyclic loads such as deadweight and residual stress are still needed for cyclic fatigue crack growth because they raise the mean stress which affects the stress distribution in the nozzle and weld.

Request:

a) Discuss why deadweight load and residual stress are not included in the finite element model to calculate stresses.
b) Discuss why seismic loads are not included in the finite element stress analysis.
c) Discuss whether there are any other loadings applied to the feedwater nozzle and nozzle-to-shell welds besides the pressure load and thermal load in the stress analysis.

ENERGY NORTHWEST RESPONSE TO RAI 2.6:

a) Deadweight load (353 kip-in) was not considered because it is relatively small compared to the thermal moments (2301 kip-in) and not cyclic. Deadweight only causes a mean stress in fatigue, and in the fatigue analysis, a high R-ratio of 0.7 consistent with Section 5.4 of BWRVIP-108-A was used to account for the increase in the mean stress. This is also consistent with the examples in Sections 4.1, 4.3 and 4.4 of BWRVIP-241-A where deadweight is not included in the analyses.

   Weld residual stress (after post weld heat treatment) was included in the probabilistic fracture mechanics design input (Enclosure 2, Section 5.2.1.3) to calculate probabilities of failure. The weld stress distribution is shown in Figure 20 of Enclosure 2.

b) For vessels and associated nozzles, the seismic load cycles are relatively very small and associated with very low cycles (I.e. 50 cycles) relative to thermal transients and therefore were not included in the finite element stress analysis. This is consistent with the methodology in BWRVIP-108-A.

c) For the reasons stated in responses (a) and (b) above, no other loadings other than pressure and thermal loads were applied to the feedwater nozzle and nozzle-to-shell welds in the stress analysis.

NRC REQUEST NO. 2.7:

Issue:

NRC's safety evaluations for BWRVIP-108-A and BWRVIP-241 require that the maximum reactor vessel heatup/cooldown rate be limited to less than 115 ºF/hour.

Request:

Confirm that CGS will satisfy this condition.

ENERGY NORTHWEST RESPONSE TO RAI 2.7:

CGS's reactor vessel heatup/cooldown rate is below the 115°F/hr evaluated in BWRVIP-108-A and BWRVIP-241-A. Per CGS's Technical Specification, the reactor vessel operational heatup/cooldown rate is limited to 100°F/hr.

NRC REQUEST NO. 2.8:

Issue:

The NRC staff compared the thermal transient cycles used in the feedwater nozzle analysis as shown in Table 5 in Enclosure 2 to the thermal transient cycles used in the recirculation outlet nozzle at CGS as shown in Table 5-5 of BWRVIP-241. The NRC staff noted that the thermal transients in Table 5-5 of BWRVIP-241 are for 40 years of operation whereas the thermal transients in Enclosure 2 are for the 60-year plant life. The NRC further noted that Table 5 in Enclosure 2 does include more transient categories than that of Table 5-5 in BWRVIP-241. Nevertheless, the NRC staff identified the following three discrepancies between Table 5 in Enclosure 2 and Table 5-5 in BWRVIP-241. (1) Table 5 in Enclosure 2 does not identify the Scram transient. It does have a "other scram" category but with only 90 cycles. The Scram category in Table 5-5 in BWRVIP-241 indicates 180 cycles. (2) The "natural recirculation startup" transient shows 3 cycles in both Table 5 in Enclosure 2 and Table 5-5 in BWRVIP-241. However, it seems that Table 5 should have more cycles than that of Table 5-5 because Table 5 is for the 60-year plant life whereas Table 5-5 is for 40 years. (3) Table 5-5 in BWRVIP-241 has a "loss of feedwater pump" transient whereas Table 5 in Enclosure 2 does not. In addition, the NRC staff notes that Section 6.1 of BWRVIP-05, "BWR Vessel and Internals Project: BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," states that loss of feedwater heaters will affect feedwater nozzles in terms of fatigue. Rapid cycling fatigue was found to occur as a result of mixing of relatively colder water with the hotter reactor water, which was addressed by modifications and design changes to the feedwater nozzles.

Request:

(a) Explain these three discrepancies between the thermal transient cycles in Table 5 in Enclosure 2 and Table 5-5 of BWRVIP-241.
(b) Clarify why Enclosure 2, Table 5 does not include the loss of feedwater heaters transient.

ENERGY NORTHWEST RESPONSE TO RAI 2.8:

a) Unlike in Table 5-5 of BWRVIP-241-A which selected the bounding transients based on the steam side of the reactor pressure vessel alone, the selection of the bounding transients in Table 5 in Enclosure 2 was based on both main steam piping transients and feedwater piping transients. Recall in Section 3.1.1 of Enclosure 2 that transients were screened to obtain the limiting events that bound the other transients. The criteria for selecting the worst cases were (a) transients that had large temperature difference and/or (b) transients that had a drastic rate of change in temperature. The bounding transients were Transient 3 (Start-up), Transient 10 DN (Turbine Generator Trip), Transient 20-2 (Loss of FW Pumps, Part-2), Transient 22 (Reactor Overpressure), and Transient 26 (Improper Startup). Note that these transients were more conservative than the bounding transients in Table 5-5 of BWRVIP-241-A, as they exhibit temperature drops/step-up as shown in the table below.

Bounding Feedwater Transients	T_A, °F (steam)		T_{FW}, °F (feedwater)	
	ΔT_{max}*	$T_{max\text{-}rate}$**	ΔT_{max}*	$T_{max\text{-}rate}$**
3. Start-up	452	0.0278	220	Drop from 320°F to 200°F
10 DN. Turbine Generator Trip	538	6.5000	122.1	2.035
20-2. Loss of FW Pumps, Part-2	152	0.0650	385	Drop from 485°F to 100°F
22. Reactor Overpressure	152	15.5000	172.1	Step-up from 100°F to 250°F
26. Improper Start-up	452	0.0278	462	Drop from 552°F to 90°F

Note: T_A, main steam temperature,
T_{FW}, feedwater temperature
* maximum temperature change of the transient
** maximum rate of change in temperature, °F/second.

In Table 5 in Enclosure 2, a conservative approach was applied to combining the cycles of similar transients (See Tables 1 through 4 of Enclosure 2 for all transient definitions) into the five bounding feedwater transients from the table above:

Transient 3 bounds Transients 2 and 4.
Transient 10 DN is evaluated alone.
Transient 20-2 is evaluated alone.
Transient 22 bounds Transient 23.
Transient 26 bounds Transient 27.

However, in Table 5 of Enclosure 2, there were also other transients that had relatively higher cycles and were much less severe than the five bounding feedwater transients in the table above. These high cycle, less severe transients were more realistically bounded by two additional representative transients:

Transient 6 bounds Transients 5 and 7.
Transient 14 bounds Transient 8, 9, 11, 13, 15 – 18, 21, and 28.

Transient 6 was selected since it had the larger feedwater temperature change as compared to Transients 5 and 7. Transient 14 was selected since it had the larger steam temperature change as compared to the other represented transients.

1. NRC Issue: Table 5 in Enclosure 2 does not identify the Scram transient. It does have a “other scram” category but with only 90 cycles. The Scram category in Table 5-5 in BWRVIP-241 indicates 180 cycles.

   Energy Northwest Response: The “Other scram” (Transient 11) was included in the transient combination of Transient 14 (see Enclosure 2, Table 5, Column 4: Bounding Transient) which had 1491 total number of cycles. In addition, Transient 14 in Table 5 in Enclosure 2 had a larger maximum temperature change (ΔTmax) and a larger maximum rate of change in temperature (Tmax-rate) compared to the “Scram” transient of Table 5-5 of BWRVIP-241-A.

2. NRC Issue: The “natural recirculation startup” transient shows 3 cycles in both Table 5 in Enclosure 2 and Table 5-5 in BWRVIP-241. However, it seems that Table 5 should have more cycles than that of Table 5-5 because Table 5 is for the 60-year plant life whereas Table 5-5 is for 40 years.

   Energy Northwest Response: The number of cycles of “natural recirculation startup” (a.k.a. natural circulation startup) transient for the 60-year plant life is 3 cycles. Reactor Startup on Natural Circulation was an original design feature for the station but it is not allowed by Technical Specifications. This upset condition is listed on General Electric (GE) Reactor Vessel Thermal Cycle Curves drawing 762E120 sheet 2 and is therefore considered in the design basis events. Since it is prohibited by technical specifications, there is no increase in the number of presumed occurrences from 40 to 60 years of plant life.

3. <u>NRC Issue</u>: Table 5-5 in BWRVIP-241 has a “loss of feedwater pump” transient whereas Table 5 in Enclosure 2 does not.

   <u>Energy Northwest Response</u>: Transient 20-2 Loss of FW Pumps, Part-2 listed in Section 3.1.1 and Table 5 of Enclosure 2 was a bounding transient that was considered in the evaluation.

b) The “loss of feedwater heater” transient was not listed in Table 5 of Enclosure 2. However, it was represented by two other transients: Transient 8 “turbine trip from 100% load with 25% bypass” and Transient 9 “partial feedwater heater bypass.” These transients were bounded by the transients discussed in Section 3.1.1 of Enclosure 2.

NRC REQUEST NO. 2.9:

<u>Issue:</u>

Section 4.1.3 of BWRVIP-108 discusses various pressure loads where Load Case 1 represents the end cap pressure load which is applied on the reactor vessel and on the nozzle. Load Case 2 is where an axial load of 1 kips is applied to the safe end of the nozzle. Load Case 3 is the in-plane bending moment applied at the nozzle (perpendicular to the RPV centerline). Load Case 4 is the out-of-plane bending moment (parallel to RPV centerline). Section 4 in Enclosure 2 appears to model these load cases such as the end cap pressure load. However, it is not clear whether Enclosure 2 modeled the in-plane and out-of-plane bending moments on the feedwater nozzle.

<u>Request:</u>

Clarify whether these four load cases are applicable to the CGS feedwater nozzle. If they are applicable, discuss whether these four load cases are included in the finite element analysis for the CGS feedwater nozzle.

<u>**ENERGY NORTHWEST RESPONSE TO RAI 2.9:**</u>

The load cases applied in the finite element analysis of the feedwater nozzle are unit moment loads in the orthogonal directions. This is described in the response to RAI 2.2. Axial forces were not included because their effect on the stresses in the nozzle and vessel are relatively small compared to the moments. For this reason, in ASME Code Section III, NB-3600 axial loads are not included in determination of the piping stresses.

NRC REQUEST NO. 2.10:

Issue:

It seems that the deterministic fracture mechanics (DFM) evaluation and some its results are not discussed in Enclosure 2. The NRC staff noted that the DFM evaluation should be part of the probabilistic fracture mechanics (PFM) evaluation to demonstrate the acceptability of reducing inspection of number of the feedwater nozzles and nozzle-to-shell welds.

Request:

(a) Discuss the equations, input values, and calculations for the DFM evaluation, including initial flaw size, crack growth, final flaw size, applied stress intensity factor, and material fracture toughness that were used for CGS.
(b) Discuss how the axial flaw and circumferential flaw are modeled in the nozzle radius and nozzle-to-shell weld in the DFM evaluation for CGS. Specifically, discuss the initial flaw size and the direction of the axial and circumferential flaw propagation in the nozzle radius and nozzle-to-shell weld.
(c) The results of the DFM evaluation at the end of the plant life should be in terms of (c1) the final size of the postulated axial and circumferential flaws in the nozzle inner radius and nozzle-to-shell weld, and (c2) the applied stress intensity factor (Ki) of the final flaw size as compared to fracture toughness of the material (K_{IC}).
(d) Clarify whether the weld residual stress was calculated in the deterministic stress analysis in Enclosure 2, Section 4.0. If not, provide justification.
(e) Use the equations in the DFM evaluation and the algorithm flow diagram in the PFM evaluation to describe two runs (two realizations) of Monte Carlo simulation to show how the PFM calculation is performed from the flaw initiation, flaw growth, to the probability of failure.

ENERGY NORTHWEST RESPONSE TO RAI 2.10:

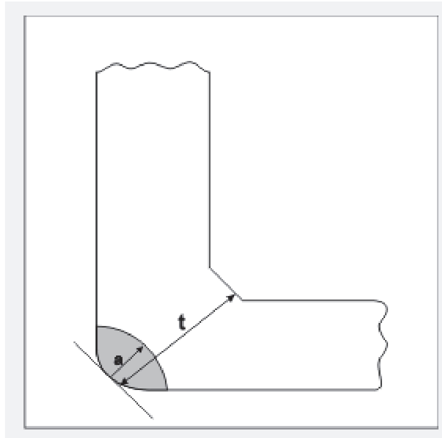
a) Each PFM simulation in VIPER-NOZ involved a DFM run. A confirmatory DFM evaluation is performed outside of VIPER-NOZ using average parameters similar to what was done in Section 6 of BWRVIP-108-A. The mean parameters in Table 8 of Enclosure 2 are used for the equation of the crack growth (SCC and FCG), SCC threshold (10 ksi√in), FCG threshold (0 ksi√in), and fracture toughness (K_{IC} = 200 ksi√in). The evaluation was performed using pc-CRACK 4.3, which is a computer software verified under Structural Integrity Associates' Nuclear Quality Assurance program for fracture mechanics evaluation. The software was used to perform crack growth analyses. Automated in the program, pc-CRACK 4.3 applies API-579-1 Fitness-for-service for the calculation of the stress intensity factors for the crack growth analysis.

   Given that there are six stress paths evaluated, crack growth analyses are performed for all six locations with each location representing one realization. Hence, a total of six realizations are performed. An initial flaw depth of 0.15 inch is assumed for all crack configuration, consistent with Section 6 of BWRVIP-108-A. For axial and circumferential crack, a 1:6 ratio is assumed as crack depth to crack length ratio, which means that the initial crack length is 0.90 inch.

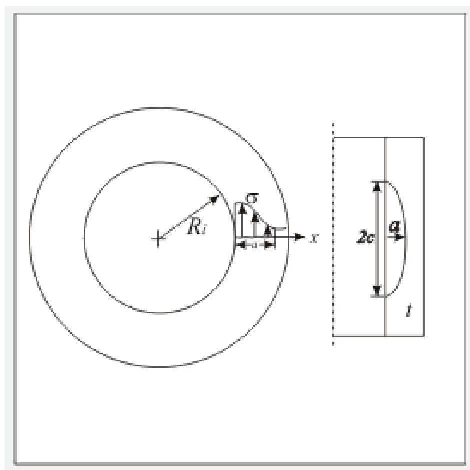
   Similar with the PFM analyses, the stresses that are described in Sections 4 of Enclosure 2 is used in the DFM analyses.

b) For path locations P1 and P2, the model is semi-circular nozzle corner crack with a given initial depth (See in the figure below). The cracks are modeled to propagate from the inside blend radius at the smallest distance between the inner and outer blend radius.

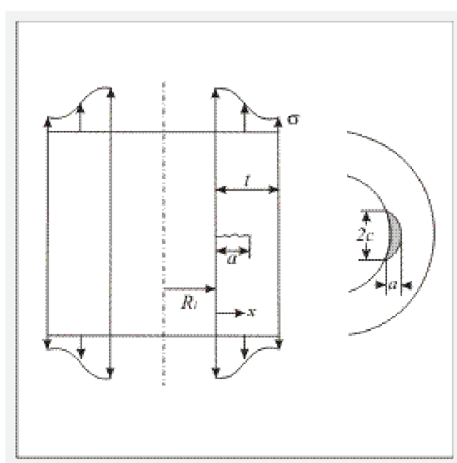
   The crack-models are semi-elliptical axial crack for path locations P3, P4, and P5 while the crack model is a semi-elliptical circumferential crack for path location P6. They are developed from the inside surface of a cylinder with a given initial depth and length. The cracks are propagated through the cylinder wall thickness from the inside radius to outside radius in depth and length with a fixed aspect ratio. Refer to the figure below for the axial and circumferential crack models used in pc-CRACK 4.3.



Nozzle Corner Crack Model from pc-CRACK 4.3



Semi-elliptical Axial Crack in a Cylinder from pc-CRACK 4.3



Semi-elliptical Circumferential Crack in a Cylinder from pc-CRACK 4.3

c) The table below summarizes the final depth (a_f), the final depth-to-thickness ratio (a_f/t), and final applied stress intensity factor (K_i) after 60 years of plant operation from the deterministic fracture mechanics analysis. Crack growth is not significant after 60 years of plant operation, and the final applied K_i's are below the fracture toughness of the material (K_{IC}=200 ksi√in).

Locations	Paths	Final Flaw Depth		Final K_i (ksi√in)
		a_f (inch)	a_f/t	
Nozzle Blend Radius	P1	1.1714	0.1172	105.3084
	P2	0.1978	0.0199	28.4001
Nozzle Inside Radius to Outer Blend Radius	P3	2.1748	0.3052	129.7243
	P4	0.4311	0.0605	58.1657
Vessel Shell to Nozzle Weld	P5	0.5042	0.0747	60.5237
	P6	0.3102	0.0459	29.0717

d) The weld residual stress is used in the deterministic fracture mechanics analysis for stress paths 5-6 in the nozzle-to-shell welds. The same stress coefficients of the weld residual stress discussed in Section 5.2.1.3 and Figure 20 of Enclosure 2 are used in the probabilistic fracture mechanics evaluation.

e) The example problem described in Items (a) through (c) above illustrates six DFM realizations. Each DFM realization uses different inputs for each of the six stress paths. In this example DFM evaluation illustration, the average parameters listed in Table 8 of Enclosure 2 are used for the crack growth calculation. As the results show, for this particular case, failure was not achieved at any of the locations since the maximum applied stress intensity factors were below the fracture toughness (no ruptures) and the final flaw depths were less than 80% of thickness (no leaks). In the PFM analysis, the same mean parameters are used in the evaluation but with an applied standard deviations and distribution to vary for each set of realization. The flow diagram on how the PFM runs in each realization is detailed in RAI 3.1(a). The number of failures that occur in the total realizations at each location are then used to determine the probability of failure.

3. Probabilistic Fracture Mechanics Analysis

NRC REQUEST NO. 3.1:

Issue:

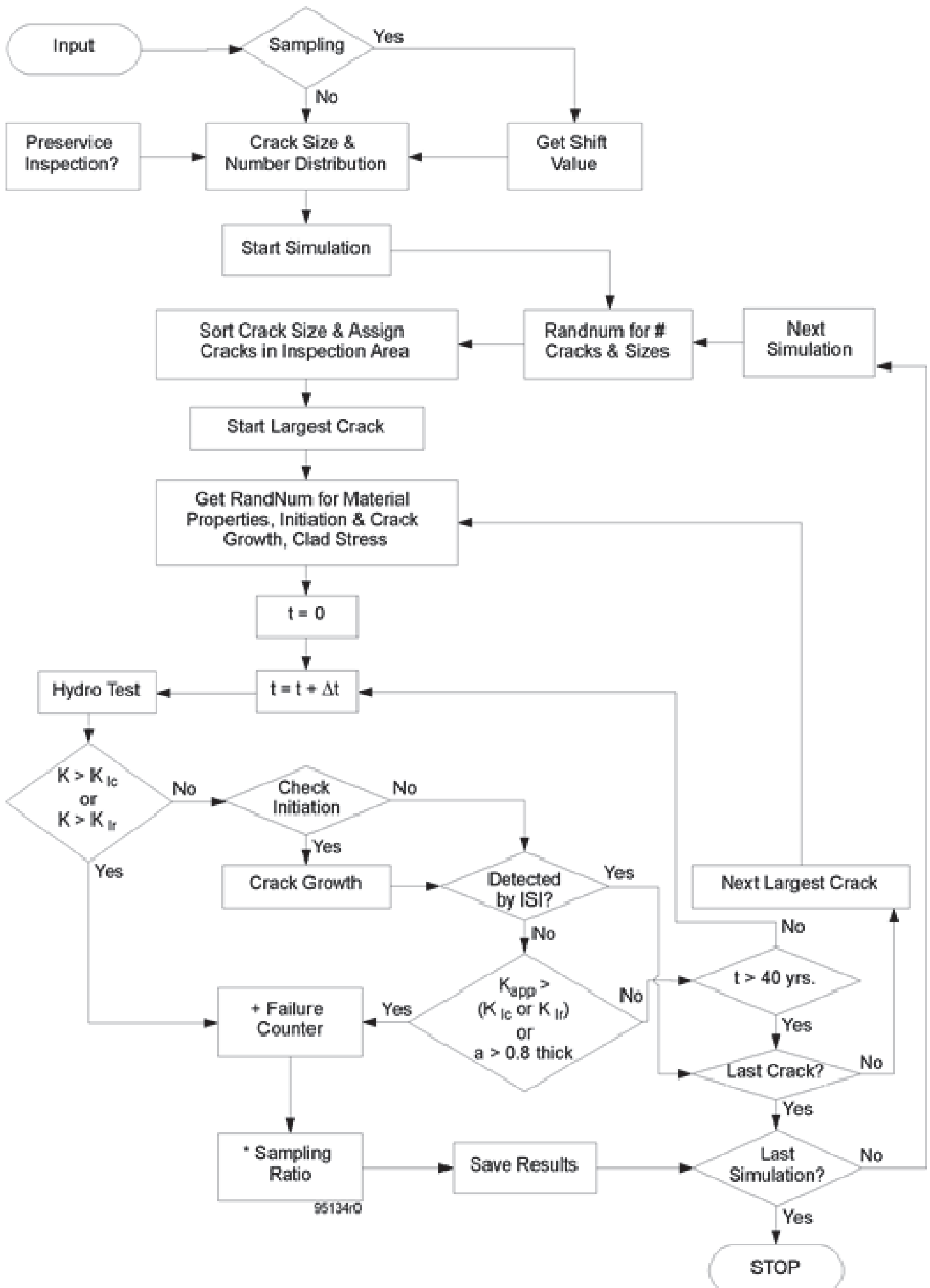
The NRC staff notes that Figure 8-1 of BWRVIP-05 provides an overview of the PFM analysis methodology. Figure 8-11 of BWRVIP-05 provides a flow diagram of the computer code VIPER. Enclosure 2 states that it follows the PFM methodology in BWRVIP-05, BWRVIP-108-A, and BWRVIP-241-A. However, it is not clear to the NRC staff exactly how the PFM evaluation in Enclosure 2 was performed.

Request:

(a) Discuss whether the PFM evaluation in Enclosure 2 is similar to the flow diagrams in BWRVIP-05. If there are differences, provide a flow diagram that explains the methodology in the PFM evaluation in Enclosure 2.
(b) Discuss how the stress paths in the stress analysis are used to derive the applied stress intensity factors.

ENERGY NORTHWEST RESPONSE TO RAI 3.1:

a) The PFM evaluation in Enclosure 2 follows the VIPER-NOZ flow diagram below, which is similar to the flow diagram in BWRVIP-05.



b) In the fracture mechanics analyses in Enclosure 2 using the VIPER-NOZ code, the through-wall stresses for each path was input into the fracture mechanics models for the nozzle-to-shell weld or the nozzle corner crack (as appropriate), which are described in Section 5.4 of BWRVIP-241-A and have been approved by the NRC.

NRC REQUEST NO. 3.2:

Issue:

The licensee stated that its feedwater nozzle analysis was based on BWRVIP-108 and BWRVIP-241. Enclosure 2 uses the VIPERNOZ computer code to perform the PFM evaluation. The NRC staff noted that both BWRVIP reports do not analyze feedwater nozzles and associated nozzle-to-shell welds. Therefore, it is not clear how the two BWRVIP reports are applicable to the CGS feedwater nozzle analysis.

Request:

Discuss whether the version of VIPERNOZ used in the PFM evaluation for CGS is the same version as was used in the two BWRVIP reports. If not, provide the differences and discuss the impact of these differences on the CGS feedwater nozzle analysis.

ENERGY NORTHWEST RESPONSE TO RAI 3.2:

The same Version 1.1 of VIPER-NOZ was used in the evaluation in Enclosure 2 and the evaluations contained in BWRVIP-108-A and BWRVIP-241-A.

NRC REQUEST NO. 3.3:

Issue:

Enclosure 2 does not discuss how the flaw growth due to stress corrosion cracking (SCC) is combined with the flaw growth due to fatigue to calculate the probability of failure of the feedwater nozzle and nozzle-to-shell weld.

Request:

Clarify how the SCC flaw growth is added to the flaw growth due to fatigue in the calculation of probability of failure of the feedwater nozzle and nozzle-to-shell weld.

ENERGY NORTHWEST RESPONSE TO RAI 3.3:

SCC crack growth and fatigue crack growth are calculated independently and summed in succession. At each time step, SCC crack growth is calculated and added first, and then, fatigue crack growth is added to the flaw size after SCC crack growth. Within the

PFM methodology (See flow chart in response to RAI 3.1 (a)), flaw detection and vessel failure are checked after crack growth from both SCC and fatigue.

NRC REQUEST NO. 3.4:

Issue:

Enclosure 2, Section 5.1, Item 6 indicates that K_{IC} of the feedwater nozzle is 200 ksi√in. The NRC staff noted that K_{IC} of 200 ksi√in is applicable to material at high temperature. K_{IC} at lower temperature would be lower than 200 ksi√in.

Request:

Discuss whether the temperature at the feedwater nozzle and nozzle-to-shell weld stays at sufficient high temperature at the time of maximum total applied load to qualify for the use of 200 ksi√in.

ENERGY NORTHWEST RESPONSE TO RAI 3.4:

Figures 13 through 19 of Enclosure 2 show the maximum stresses during the seven bounding transients listed in Table 5 of Enclosure 2. Comparing the figures for all bounding transients, the highest stresses (49 ksi) occur during Transient 3 (Start-up) at Stress Path 4, as shown in Figure 13 of Enclosure 2. Among all stress paths in the evaluation, the minimum temperature at the inside pipe surface is 280 ^{o}F (Stress Path 4) at the time of maximum total applied load during Transient 3 (20072 seconds).

The maximum RT_{NDT} of the six feedwater nozzles is 0 ^{o}F, and the maximum RT_{NDT} among the three RPV shell pieces at the feedwater nozzles region is 12 ^{o}F. The feedwater nozzles are far remote from the beltline region and therefore these initial RT_{NDT} values do not have to be adjusted for the effect of fluence.

The minimum (T- RT_{NDT}) is calculated as 280 – 12 = 268 ^{o}F.

Per Figure A-4200-1 of ASME Code, Section XI, Appendix A, the value of K_{Ic} is 220 ksi√in at the time of maximum total applied load for the CGS Feedwater Nozzle. Hence, the lower value of 200 ksi√in used in the evaluation is conservative.

NRC REQUEST NO. 3.5:

Issue:

Enclosure 2, Section 5.0 states that for the nozzle-to-shell weld, either a circumferential or an axial crack, depending on weld orientation, can initiate due to either component fabrication (i.e. considering only welding process) or stress corrosion cracking. It is generally known that welds get repaired during construction of nuclear plants. The operating experience has shown that construction repair creates weld residual stresses which could increase the probability of flaw initiation and growth. Enclosure 2, Section 5.0 does not discuss whether the nozzle-to-shell weld had been repaired during the original construction

Request:

Discuss whether any of the feedwater nozzle-to-shell welds was repaired at the time of the construction. If yes, discuss the flaw size that was repaired and discuss whether the weld residual stress analysis includes such repaired flaw. If not included, provide justification.

ENERGY NORTHWEST RESPONSE TO RAI 3.5:

As documented in CGS's Final Safety Analysis Report (FSAR) Section 5.3.3.3, "The shells and vessel heads were made from formed low alloy steel plates and the flanges and nozzles from low alloy steel forgings." The section goes on to state; "Postweld heat treatment of 1100°F minimum was applied to all low alloy steel welds." This heat treatment should reduce residual stresses from any repairs such that they would not be a dominant force requiring consideration in the analysis. Weld residual stress (after post weld heat treatment) was included in the probabilistic fracture mechanics design input as discussed in RAI Response 2.6 above.

NRC REQUEST NO. 3.6:

Issue:

Section 5.2.2.3 in Enclosure 2 states that the stress corrosion cracking (SCC) initiation data was based on cast stainless steel. The reactor vessel and feedwater nozzle are made of low alloy steel; therefore, the SCC initiation law should be based on the cracking data in low alloy steel.

Request:
Clarify why the SCC initiation data for cast stainless steel was used for the crack growth in the feedwater nozzle and nozzle-to-shell weld even though both components are not made of cast stainless steel.

ENERGY NORTHWEST RESPONSE TO RAI 3.6:

The SCC initiation values used in Enclosure 2, (Section 5.2.2.3 and Table 8) were selected to be consistent with the NRC safety evaluation report (SER) for BWRVIP-108-A. In BWRVIP-108 SER (See Page 9), sensitivity studies were performed in response to RAI 2-10. Case 2 examined a lower coefficient of SCC initiation time of 0.842×10^{20} versus 4.21×10^{20} used in BWRVIP-108. The NRC staff concluded that "the combined effect of Cases 1 to 7 should be evaluated because they, collectively, reflect the relevant material property for forging and represent established positions regarding PFM analysis inputs as a result of the BWRVIP-05 review." In the subsequent analysis to address the combined effect of all the parameters, the BWRVIP used the SCC initiation time coefficient of 0.842×10^{20}, as requested, with acceptable results. As noted in BWRVIP-05, low allow steel components such as CGS's feedwater nozzles are highly resistant to SCC initiation in boiling water reactor environments so use of the cladding initiation coefficient is considered conservative.

NRC REQUEST NO. 3.7:

Issue:

Enclosure 2, Section 5.2.2.5 discusses the comparison between the fatigue crack growth data that were used in the CGS analysis and the fatigue crack growth law in the ASME Code, Section XI in a reactor water environment. The licensee stated that its fatigue crack growth data show a reasonable comparison; however, the fatigue growth law in the ASME Code, Section XI is more conservative than that in the EPRI report on growth rate at high ΔK ($K_{max} - K_{min}$).

Request:

Discuss whether the fatigue crack growth curves used in the CGS PFM evaluation is adequate when the fatigue crack growth curve at high ΔK ($K_{max} - K_{min}$) in the ASME Code, Section XI is more conservative than the fatigue crack growth curves used in the PFM evaluation.

ENERGY NORTHWEST RESPONSE TO RAI 3.7:

The fatigue crack growth curve used in the evaluation in Enclosure 2 is consistent with that used in BWRVIP-108-A and BWRVP-241-A, which were approved by the NRC.

NRC REQUEST NO. 3.8:

Issue:

It appears that the POD in Enclosure 2, Table 10 is applied to both the feedwater nozzle radius and nozzle-to-shell welds. The NRC staff noted that the nozzle radius and nozzle-to-shell weld are of different shape and thickness. The ultrasonic examination coverage may be different between the feedwater nozzle radius and the nozzle-to-shell weld.

Request:

Justify why the same POD is applicable to both components.

ENERGY NORTHWEST RESPONSE TO RAI 3.8:

The POD values used in the PFM are identical to the POD values used in BWRVIP-108-A, Figure 2-1 for Pass Plus Fail. The BWRVIP-108-A curve considers both nozzle-to-shell welds and inner radii data. This is clear from the discussion in BWRVIP-108-A, Section 2 and the BWRVIP-108-A SER, Section 4.2. Therefore, using the same POD for both the nozzle-to-shell welds and inner radii is considered appropriate for the PFM. As noted in RAI 1.2 all of the components received greater than 99% coverage.

NRC REQUEST NO. 3.9:

Issue:

Enclosure 2 does not describe the interface between the probability of detection (POD) and the inspection of 25% of the six feedwater nozzles.

Request:

Discuss how the POD is combined with the 25% inspection (i.e., inspecting 25% of the six feedwater nozzles) is analyzed to reach the probably of failure.

ENERGY NORTHWEST RESPONSE TO RAI 3.9:

There is no correlation between the percentage of inspection and the POD curve. The POD curve characterizes the effectiveness of the inspection method. The POD curve and inspection percentage are two separate, independent inputs into VIPER-NOZ. The PFM methodology in VIPER-NOZ will first consider whether a nozzle is inspected (inspection percentage) and then whether a flaw is detected (POD curve) at each time step.

NRC REQUEST NO. 3.10:

Issue:

The NRC staff notes that with the proposed 25% of inspection of the six feedwater nozzles, there will be feedwater nozzles that will not be inspected for the remainder of the plant life. In this scenario, the PFM evaluation would need to assume that there is no inspection for those feedwater nozzles and calculate the probability of failure accordingly (i.e., 0% inspection, not 25% inspection).

Request:

Discuss whether this is a plausible scenario for the PFM evaluation. If yes, discuss the impact of this scenario on the final probability of failure.

ENERGY NORTHWEST RESPONSE TO RAI 3.10:

In BWRVIP-108-A (Section 5.7), PFM evaluations were performed with inspection sampling of 0% (no inspection), inspection sampling of 25% and inspection sampling 99% sample with the results presented in Tables 5-4, 5-5, and 5-6, respectively. In all cases, the PoF are below the acceptance criteria. Considering the low PoF values for the 25% sample in Enclosure 2 (on the order of $1x10^{-11}$), similar conclusions will be reached.

NRC REQUEST NO. 3.11:

Issue:

Enclosure 2 does not provide failure criteria for the feedwater nozzle and nozzle-to-shell weld in the PFM evaluation. Without the failure criteria, it is not clear how the probability of failure is estimated.

Request:

Clarify what are the criteria (in terms of nozzle leakage or fracture) for the failure to occur at the feedwater nozzle inner radius and associated nozzle-to-shell weld in the PFM evaluation and whether the criteria are consistent with that of BWRVIP-108 and BWRVIP-241.

ENERGY NORTHWEST RESPONSE TO RAI 3.11:

For each simulated vessel, the failure criteria within VIPER-NOZ are:

Rupture: $K_{applied} > K_{Ic}$,
where $K_{applied}$ is the applied stress intensity factor and K_{Ic} is the fracture toughness

Leakage: crack depth > 80% of vessel wall thickness

The probability of failure is defined as the number of failed vessels divided by the total number of simulated vessels. The probability of failure is compared to the acceptance criteria listed in Section 2 of Enclosure 2, which is consistent with that used in BWRVIP-108-A and BWRVIP-241-A.

NRC REQUEST NO. 3.12:

Issue:

Enclosure 2, Section 6 states that the probability of failure (PoF) for the CGS feedwater nozzles or the nozzle-to-shell welds at 25% inspection in 60 years for the normal operation condition and LTOP event are less than 1.67E-8 and 1.67E-11, respectively. However, these two PoF estimated are much less than the PoF estimated for the recirculation nozzles for a 25% inspection in 40 years in BWRVIP-108 and BWRVIP-241. For example, Table 4, Case 5 in the BWRVIP-108 Supplement dated September 13, 2007 (ADAMS Accession No. ML072600173, proprietary) shows a higher PoF for the recirculation nozzles than the CGS feedwater nozzles. Also, the PoF estimated for the CGS feedwater nozzles during the LTOP event in Enclosure 2 is less than the PoF estimated for the CGS recirculation nozzle as shown in Table 5-9 in BWRVIP-241. The NRC staff notes that the PoF for the CGS feedwater nozzles is calculated for 60 years of operation whereas the PoF for the recirculation nozzles is calculated for 40 years in BWRVIP-108 and BWRVIP-241. In general, the feedwater nozzles should have experienced more transients than the recirculation nozzles. In addition, the CGS feedwater nozzles having more applicable transients as shown in Enclosure 2, Table 5 than the corresponding nozzles analyzed in the September 13, 2007 supplement of BWRVIP-108 and Table 5-9 of BWRVIP-241. Therefore, it appears that the CGS feedwater nozzle PoF should be higher, not lower, than the recirculation nozzle PoF.

Request:

Justify why the CGS feedwater nozzles in Enclosure 2 have lower PoF than the recirculation nozzles in the BWRVIP-108 supplement and BWRVIP-241.

ENERGY NORTHWEST RESPONSE TO RAI 3.12:

The probabilities of failure for the CGS recirculation outlet nozzle reported in Table 5-9 of BWRVIP-241-A are the conditional probabilities of failure. To calculate the probability of failure due to an LTOP event, these conditional probabilities are multiplied by the probability of an LTOP event occurrence ($1x10^{-3}$). For normal operating condition, the probability of failure is the same as the conditional probability of failure. For no failure (NF) results without tabulated values, the conditional probability of failure is calculated as less than 1 failure in the total number of simulations (<1 failure / 1,000,000 simulations / 40 years = 2.5×10^{-8}). The total number of simulations is assumed to be one million, which is consistent with recirculation outlet nozzle simulations in Table 5-4 of BWRVIP-108-A.

The table below compares the probability of failure for an LTOP event and the probability of failure for normal operation at the nozzle blend radius and the nozzle-to-shell welds for the CGS recirculation outlet nozzle from Table 5-9 of BWRVIP-241-A and the CGS feedwater nozzle in Table 11 of Enclosure 2. The number of failures in the simulations are also provided in the table, and both evaluations performed one million simulations.

As seen in this table, for the nozzle-to-shell welds, there were no failures in the recirculation outlet nozzle but three failures for axial and circumferential flaws due to an LTOP event in the feedwater nozzle. This clearly shows that the probability of failure for the feedwater nozzle-to-shell welds is comparable on the same order of magnitude as that for the recirculation nozzle-to-shell welds, and the probability of failure for no failure cases is dependent on the total number of simulations. The slightly higher probability of failure of the recirculation outlet nozzle inside radius section compared to the feedwater nozzle inside radius section is attributed to the conservative combination of the transients and associated cycles in the BWRVIP-241-A evaluation of the recirculation outlet nozzle.

Component	**Crack Model**	**a/l**	**Flaw Density**	**Condition**	**Number of Failures (1 million simulations)**		**Probability of Failure (per year)**	
					Recirculation Outlet	**Feedwater**	**Recirculation Outlet**	**Feedwater**
Nozzle Inside Radius	Blend Radius	NA	0.1	LTOP	Not specified	NF	$6.83x10^{-9}$	$<1.67x10^{-11}$
				NOP	NF	NF	$<2.5x10^{-8}$	$<1.67x10^{-8}$
Nozzle-to-Shell Weld	Axial	1/2	1	LTOP	NF	3	$<2.5x10^{-11}$	$5.0x10^{-11}$
				NOP	NF	NF	$<2.5x10^{-8}$	$<1.67x10^{-8}$
	Circ.	1/2	1	LTOP	NF	3	$<2.5x10^{-11}$	$5.0x10^{-11}$
				NOP	NF	NF	$<2.5x10^{-8}$	$<1.67x10^{-8}$

NF = No failure in the simulations
NOP = Normal operating condition

Enclosure 3

Corrected page 13 for the non-proprietary version 1801567.301, Revision 1

5.2 Design Inputs

Section 5.2.1 presents the inputs modeled deterministically as constants, and Section 5.2.2 describes the probabilistic inputs considered to be random variables.

5.2.1 Deterministic Parameters

Table 9 summarizes the CGS FW dimensions from the nozzle drawings [13, 14] and normal operating conditions from the thermal cycle diagrams [15]. The LTOP event pressure and temperature are specified in the BWRVIP-05P Safety Evaluation [17].

More detailed input parameters used for inservice inspection (ISI) interval, fatigue cycles, and stress distributions are described in the following sections.

5.2.1.1 In-Service Inspection

25% inspection for 60 years is used. The probability of detection (PoD) distribution function associated with inspection is shown in Table 10 [18].

5.2.1.2 Stresses

Stresses due to internal pressure and thermal transients were determined in the stress analysis [11]. Coefficients of linearized stress paths are obtained for the PFM analysis.

5.2.1.3 Weld Residual Stresses

Consistent with BWRVIP 108-A [3], the weld residual stresses (WRS) are assumed present in the nozzle-to-shell welds for Paths 5 and 6. The WRS distribution at the nozzle/shell weld is assumed to be a cosine distribution through the wall thickness with 8 ksi mean amplitude and 5 ksi standard deviation (see Figure 20). No WRS is present in the nozzle blend radius region for Paths 1 and 2 and nozzle inside surface to the outer blend radius Paths 3 and 4.

5.2.2 Random Variables

Random variables used in the N-702 evaluation are summarized in Table 8. More detailed input parameters used for SCC Initiation, SCC Growth and fatigue crack growth are described in the following sections.

5.2.2.1 Material Chemistry

Table 8 presents the weld chemistries (%Cu and %Ni) along with the standard deviation and distributions for the nozzle forging and the nozzle-to-vessel welds, which are from BWRVIP-108-A Safety Evaluation [3].

5.2.2.2 Fluence

The CGS FW nozzle (N4) experiences fluence {{[redacted]}} n/cm^2 for 60 years of plant operation [20]. As such, a bounded fluence of 1.00E+17 n/cm^2 is used for the evaluation of the feedwater nozzles.

Enclosure 4

Affidavit for Withholding

Structural Integrity Associates, Inc.

October 12, 2020

Attention:
Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the Following Commercial Document:

1) SI Calculation No. 1801567.301P, Revision 1

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the information identified in the enclosed Affidavit consisting of the commercial information owned by Structural Integrity Associates, Inc. ("SIA") identified above (the "Calculations"). Copies of the Calculations and the Affidavit in support of this request are enclosed.

SIA desires to disclose the Calculations with proprietary information identified with brackets, {{ }} ({{ This sentence is an example.}}), in confidence to fulfill commitments outlined in Energy Northwest Contract 355228 (SIA Project 1801567.00). The Calculations are not to be divulged to anyone outside of the NRC nor shall any copies be made of the Calculations provided herein.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (408) 833-7357. Questions on the content of the Calculations should be directed to me as well.

Sincerely,

Kevin Wong

Kevin Wong
Project Manager
Structural Integrity Associates

AFFIDAVIT

October 12, 2020

RE:Request for Withholding of the Following Commercial Documents:

1) SI Calculation No. 1801567.301P, Revision 1

I, Kevin Wong, being duly sworn, depose and state as follows:

I am the Project Manager at Structural Integrity Associates, Inc. whose principal office is located at 5215 Hellyer Ave., Suite 210, San Jose, CA 95138 ("SIA") and I have been specifically delegated responsibility for the above-listed Calculations that is sought under this Affidavit to be withheld (the "Calculations"). I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Calculations on behalf of SIA.

SIA requests that the Calculations with proprietary information identified within brackets, {{ }} ({{ This sentence is an example.}}), be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Calculations are owned by SIA and constitutes commercial information which has not been placed in the public domain by SIA.

b. SIA made a substantial economic investment to develop the Calculations and, by prohibiting public disclosure, SIA derives an economic benefit. The Calculations are entitled to the protection of the United States copyright laws. If the Calculations were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry at no cost, these entities would be able to use the Calculations for their own commercial benefit and profit and without expending the substantial economic resources required of SIA to develop the Calculations.

c. SIA made a substantial investment of both money and employee hours over an extended period of time in the development of the Calculations. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Calculations are highly valuable to SIA.

d. A public disclosure of the Calculations would be highly likely to cause substantial harm to SIA's competitive position. If a party does contract with SIA to obtain the information contained herein, it would require an investment of money, time and effort equivalent to that expended by SIA for the party to duplicate the Calculations.

Sincerely,

Kevin Wong

Kevin Wong
Project Manager
Structural Integrity Associates